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Abstract

LIQUID-METAL HEAT TRANSFER

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This paper will summarize, and indicate the significance of, recent advances in the study of liquid-metal heat transfer, with particular emphasis on applications to the design of liquid-metal-cooled nuclear power plants. The paper will treat only that pertinent information generated in the United States in the years 1958 through 1963. In the second half of this six-year period, more liquid-metal heat transfer information was published than in any previous three-year period. The tendency in the United States has been to carry out both fundamental analytical and experimental studies on simple, well-defined, and controlled systems rather than to measure heat transfer rates on prototype heat-exchange equipment.

For single-phase liquid-metal heat transfer, new theoretical and semi-empirical expressions have become available for predicting heat-transfer coefficients, under various flow conditions, for both flow through channels and flow past submerged bodies. In the former category, the cases treated have been annuli, parallel plates, and in-line flow through rod (or tube) bundles; in the latter, spheres, and flow normal to cylinders of circular and elliptical cross sections. These last cases can be considered as good approximations of those involving flow at various angles across rods in a rod bundle (or tubes in a tube bank).

In channel flow, theoretical equations for Nusselt numbers contain the term ψ, the ratio of the eddy diffusivity of heat transfer to that of momentum transfer, while semi-empirical equations contain $\bar{\psi}$, an effective average value of ψ. A major problem in predicting heat-transfer coefficients is the evaluation of these ratios. We now have a better understanding of the general shape of the typical Nusselt number vs. Peclet number curve for

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Abstract

THERMAL-DESIGN ASPECTS OF GAS-COOLED REACTORS

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Air, nitrogen, and helium are commonly used as coolants in U.S. gas-cooled reactors. Core power densities range from 2 KW/liter for the Experimental Gas-Cooled Reactor (EGCR) and 8-10 KW/liter for High Temperature Gas-Cooled Reactors (HTGR) to about 250 KW/liter in a preliminary design of a fast gas-cooled reactor. The ratio of pumping power to thermal power usually lies between 2% and 5%. Core outlet temperatures range from 1050F in the EGCR to 1400F in HTGR's. Fuel element surface and internal temperatures are in the range where heat transfer by radiation must be considered. Metallic-clad fuel elements have been used with uranium oxide dispersed in thin concentric cylindrical sheets or with uranium oxide in rod clusters. Graphite-clad fuel elements with uranium carbide fuel are used in HTGR's and pebble-bed reactors.

Heat-transfer experiments have been used extensively to determine local and average transfer coefficients for all these types of fuel elements. Mass-transfer experiments with naphthalene or benzoic acid have been used to determine heat-transfer distributions in complex situations. The heat-transfer correlations for parallel flow seem in general to be close to the standard correlations, although local deviations can be large and must be carefully investigated for metal-clad elements. For instance, the local heat-transfer coefficient in the Experimental Beryllium Oxide Reactor (EBOR) varies 20% to 30% around its average value.

Surface roughening has been investigated as a means of increasing surface heat transfer coefficients and improving core performance. Results have shown

that the number of fuel elements in a reactor can be reduced by as much as a factor of four by roughening only an optimized fraction of the surface. It has been shown for circular tubes that swirling of the gas coolant may give heat-transfer coefficients between 1.1 and 3 times those obtained with turbulent flows. The swirl may be introduced by tangential injection or twisted tapes.

Statistical determination of hot-spot factors is used for metallic-clad gas-cooled fuel elements. In ceramic fuel elements local hot spots are less important than the fraction of the fuel above a given temperature where fission-product release increases rapidly. The problem of cross flow between coolant channels in HTGR-type reactors has been shown not to affect appreciably the thermal analysis.

Special heat-transfer and pressure-drop correlations have been obtained experimentally for pebble-bed reactors. Thermal stresses in the spheres, levitation, and nonuniform flow distribution have been investigated.

As most of the thermal-design analysis is performed by numerical methods, numerous digital computer codes have been developed to analyze 2- and 3-dimensional problems. The finite difference equations resulting from the relevant differential equations are usually made implicit to avoid time-step limitations in transient problems. The resulting set of nonlinear equations is solved by iterative methods; nonlinear coeffieients (such as temperature-dependent properties and thermal radiation) are treated by re-evaluating the coefficients, as a function of the current temperature, between iterations.

In several gas-cooled reactors, the loss-of-coolant problem limits the power density and often determines the geometry of the fuel element. The thermal stresses generated by power or coolant transients may limit the freedom of design of ceramic fuel elements.

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Abstract

PROBLEMS IN THE DESIGN AND CONSTRUCTION OF LARGE REACTOR VESSELS

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Reactor vessels for pressurized and boiling water system nuclear power plants of 500 to 600 MWe capacity are now under construction in the United States. Designs are being finalized for reactor vessels for 1000 MWe plants, with studies being made of vessels for plants up to 1500 MWe capacity. The relation between power plant size and reactor pressure vessel size is shown for the pressurized water (2500 psi design) and boiling water (1250 psi design) nuclear power systems.

The philosophy of pressure vessel design for nuclear applications is being changed. Sections I and VIII of the ASME Code required the pressure vessels to be sized on the "maximum stress theory". The new philosophy of Section III of the Code utilizes the "maximum shear stress theory". The reason for the change in philosophy and the effect of this change on the designs for large reactor vessels is discussed.

A reactor vessel for a 1000 MWe pressurized water plant designed to Section I or VIII of the ASME Code would require plate materials of about 14 in. in thickness. The vessel flange would be 230 in. in outside diameter and about 18 in. thick. A reactor vessel for a 1000 MWe boiling water plant designed to the same Code requirements would require plate materials of about 9 in. in thickness. The vessel flange would be 292 in. in outside diameter and about

15 in. thick. The problem in detailing the design of large vessels so that quality materials may be purchased for fabrication are discussed. The various grades of Code-acceptable steels are compared from economic and design standpoints for use in reactor pressure vessels.

The problems in inspection of these materials to determine that the quality meets the unique requirements for this application are described. The investigation of radiation damage to reactor vessel materials is not included in this paper. The effect of acceptance of high-strength steels, produced by quenching and tempering heat treatments, on the designs is presented together with a delineation of the problems associated with low cycle fatigue loadings.

The evolution of fabricating techniques for large, heavy vessels is reviewed. The space and handling problems presented by the increased use of reactor vessels of 15 to 22 feet in diameter, weighing up to 800 tons, is discussed. Plant siting requirements are stipulated to enable complete shop fabrication, testing, and shipment of these reactor vessels. The time required from order date to delivery is forecasted for the range of reactor vessels required by nuclear power plants of up to 1500 MWe.

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Abstract

COMPONENTS FOR SODIUM REACTORS

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Despite the many successes and advances in recent years on sodium system technology, practically every aspect still presents a challenge to the skill and ingenuity, and is amenable to substantial improvement. This paper describes the major components of sodium reactor system - reactor vessels, steam generators, heat exchangers, and pumps - and notes some problems are being worked on.

The United States is fortunate in having been able to gain an appreciable amount of experience with diverse designs of major components in sodium reactor systems since the Second Geneva Conference, 1958. The Hallam Nuclear Power Facility, a 256 MW(th), graphite-moderated sodium reactor, has been tested, commissioned, and operated for months at full power. The Experimental Breeder Reactor II, a 62.5 MW(th) fast breeder, has completed extensive pre-operational testing and has commenced power operation. The Enrico Fermi plant, a 300 MW(th) fast breeder, also has conducted pre-operational testing and is undergoing low-power operation. Smaller and older installations continue to provide information; e.g., Los Alamos Molten Plutonium Reactor Experiment (LAMPRE), Sodium Reactor Experiment (SRE), and EBR-I. In addition, experience to be gained from these systems in the future is expected to be augmented strongly by specific component tests and test programs, some currently in progress, fostered by the U. S. Atomic Energy Commission Sodium Components Development Program.

Emphasis in steam generators and heat exchangers is placed on the design

features employed to prevent contact between sodium and water and to minimize stress corrosion, thermal shock, and vibration. Failures have been experienced with a Hallam intermediate heat exchanger and a Fermi steam generator. One of six Hallam heat exchangers suffered rupture of a tube. The difficulty was traced to tube vibration induced by high velocity of sodium near an inlet. Similar vibration failed several tubes in one of the three Fermi steam ganarators, resulting in a significant sodium-water reaction. Modifications of tube supports and/or flow baffles effectively eliminated the problem. In EBR-II, no difficulty with these components has been experienced to date, but operation has barely started.

None of the vessels has failed or sustained significant difficulty to date. However, only the Hallam unit has yet experienced the principal operating conditions for which it was designed. Of greatest interest at this time are the relative complexities of these vessels, particularly in respect to thermal baffling for prevention of excessive thermal stresses, and the resultant effects upon fabrication effort.

Hallam plant pumps, of the mechanical centrifugal type, have experienced failure or operating difficulty from several causes including accumulation of foreign material within pump wear-ring clearances and misalignment of the pump casing arising from improper temperature distribution. Removal of all six pumps was necessitated for corrective action. Both EBR-II main primary pumps failed during pre-operational testing. Failure in each case was attributed to rubbing between the pump shaft and a "labyrinth seal" provided to restrict flow of sodium vapor along the shaft. Substitution of shafts less susceptible to warpage and of labyrinth seals with greater clearance is thought to have resolved the problem. Centrifugal pumps of the Fermi plant have operated completely satisfactorily for several thousand hours. In all of these plants, the sodium-lubricated (hydrodynamic) bearings of the pumps have been trouble-free.

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Abstract

IMPROVEMENTS IN INSTRUMENTATION FOR HIGH-PERFORMANCE POWER REACTORS

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Development of nuclear power plants capable of supplying low-cost power has required increased performance from the reactor core. In order that reactor performance will not be limited by instrumentation capability, new instruments have been developed, both for new functions not hitherto utilized and for improved realization of functions already used. Low power cost also requires high plant availability, and much effort has been expended in improving instrument reliability to this end. Some of the newly developed instrumentation techniques are described.

A major improvement in nuclear electronics for reactor instrumentation has been effected by the development of special-purpose modular instruments using solid-state active components. Besides the advantages of increased performance and reliability and decreased size and power requirements, these new modules can be combined to form instrument systems with greatly improved characteristics. A wide-range counting channel is described using continuous controlled motion of the sensor to obtain measurement of reactor power and period over the entire range of operation, without range switching or other discontinuities. Multiple reactor controllers have been developed in which failure of one measuring channel or one servo will not result in loss of automatic regulation of the process. Safety systems with complex input functions have been built using in-process

testing to assure adequate safety performance without sacrifice of plant serviceability.

Improved nuclear detectors described include compensated ionization chambers, multi-section chambers, high-temperature chambers, and a fission-chamber pulse-preamplifier assembly insensitive to electrical pickup noise. Non-nuclear instrumentation modules using solid-state components have been developed which give improved performance and reliability.

Special sensors are described for measuring non-nuclear variables in the reactor environment. Ultrasonic techniques are used to measure liquid level, or liquid foam vapor interfaces and foam density. Experimental sensors include sonic, ultrasonic, and reactivity detection of incipient boiling, pneumatic temperature measurement, and measurement of high temperature using variation of the dielectric constant of certain ceramics. In-core sensing of nuclear and non-nuclear variables is used to facilitate optimization of reactor performance and control.

The use of these new instruments and techniques is intended to increase plant safety, performance, and serviceability. The undoubted increase in the complexity and cost of instrumentation must therefore be offset by the above-mentioned gains, leading to lower overall cost with acceptable safety.

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Abstract

FLUID DYNAMICS, STABILITY, AND VAPOR-LIQUID SLIP IN BOILING REACTOR SYSTEMS

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Novak Zuber (General Electric Co. - Schenectady)
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Recent progress in the study of two phase flow at government laboratories, universities and in industry is reviewed.

Studies at Argonne National Laboratory cover pressure drop and slip ratio in two phase flow through nozzles, slip ratios in horizontal and vertical flow, the effect of liquid viscosity on slip ratio, slip in two component systems, slip at low mass flow rates, vapor carryover problems, and a few selected instrumentation techniques for measuring void fractions.

Work reported from the General Electric Company deals with such problems as the prediction of voids in bubbling systems, boiling in horizontal flow, the steady state and transient response of void systems, the propagation of kinematic waves in two phase mixtures, the flow stability of boiling systems and steam separation studies.

With reference to the stability of boiling systems, a review is given of experiments on natural circulation flow stability, both in industry and in various universities. Two conclusions drawn from this review are (1) flow instabilities may arise from either the boiling section of the system, or the external parts of the system, or an interaction between the boiling section and the rest of the system; (2) no definite statements related to the onset of flow instability can be made unless the region of operation of the system is carefully specified, and the direction of change of the system variables is also specified.

Two principal areas studied by Allis Chalmers in regard to reactor core hydraulics are pressure drop and steam volume fraction (void fraction). Pressure drop studies conducted at 600 psi in support of the Pathfinder reactor

project showed that the parallel-rod geometry could be treated with the established two phase pressure drop correlations. An extensive study of void fractions in flow conditions similar to those found in "risers" is being conducted under the Euratom program. A limited study of void fractions in parallel-rod geometry was conducted for Pathfinder. In conjuction with the steam separation program, void fraction measurements were also taken on bubbling two phase mixtures. These data covered a range of pressures from 300 to 2000 psi and vessel diameters from 4 to 36 inches.

Studies of boiling water reactor stability at the Space Technology Laboratories have included laboratory measurements of the power-void transfer function in a simulated SPERT IA water coolant channel. These transfer function measurements with natural circulation showed that the divergent power oscillations observed in the SPERT IA reactor at 97C can be explained as a reactivity feedback instability, not as a purely hydrodynamic instability. As the power level in this channel is raised to approach the threshold of hydrodynamic instability, a resonance peak develops in the transfer function at 1 cycle/sec, the frequency at which a hydrodynamic oscillation eventually occurs. The resonance indicates that the void response to power modulation becomes dominated by the interaction between inlet flow and steam void, and when this feedback interaction becomes unstable a spontaneous hydrodynamic oscillation occurs.

An attempt was made to reproduce in the laboratory the destructive pressure pulses which occurred in the SPERT I and BORAX I reactor excursions. Transient electrical heating produced surface temperature rises of 700C in 150μsec with a stainless steel heated element in a geometry which modelled the unmolten SPERT I and BORAX I metal cores. The magnitudes of the pressure pulses produced by heat transfer to the water were about 800 psi, which is much less than was observed in the reactor excursions. Negative gauge pressure (vacuum) follows vapor blanketing in these thermal transients.

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Abstract

HELICAL, FORCED-FLOW HEAT TRANSFER AND FLUID DYNAMICS IN SINGLE AND TWO PHASE SYSTEMS

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This paper investigates the influence of superimposed, rotational motion on forced-flow heat transfer and fluid dynamics in single and two-phase systems. Helical flow may be developed by forcing a fluid through a tube that contains a continuous, twisted divider or a helical guide vane located only at the tube wall. Under these circumstances the flow patterns are completely or partially constrained by flow guides. Helical flow may also be generated in a tube by forcing the fluid through tangencial entrance ducts. The resulting flow pattern is unconstrained, decaying rotational flow superimposed on a constant axial flow component.

A number of investigators have measured the heat transfer and fluid friction characteristics of gases and liquids under helical flow in tubes. In general, the heat transfer conductances and friction factors or pressure drops were found to increase with the magnitude of the rotational fluid flow component. The specific fluid velocity profiles and the magnitude of wall heat addition also influenced the experimental results. Two-phase or boiling heat transfer and fluid friction measurements for liquid metal and ordinary liquids under helical flow in tubes are also available. Again, heat transfer conductances and friction factors or pressure drops were found to increase with the magnitude of the rotational fluid flow component as was the case for single-phase flow; however, the functional variation is more complicated because of the various ways in which the two phases can distribute themselves in the tube.

Heat and momentum transfer in helical single-phase flow can be predicted by accounting for free and forced convection mechanisms which exist in the

curvilinear flow patterns. The free convection term for a given fluid is controlled by the acceleration field produced by the velocity field of the rotating fluid and the magnitude of the wall heat addition. The forced convection term in helical flow for a given fluid is defined by the turbulence level which normally is large because of the relatively large vectorial fluid velocities and geometrical turbulence promotion. In two-phase or boiling systems, the acceleration forces in helical flow control the spacial distribution of liquid films and droplets in boiler tubes and the corresponding heat transfer and fluid friction. In the low vapor quality region of a boiler, centrifugal forces in rotational flow force liquid to be located contiguous to the tube wall thereby yielding more effective heat transfer. In the high vapor quality fog flow region, the centrifugal forces increase the density of the liquid droplets in the vapor layers contiguous to the tube wall thereby significantly increasing the evaporative heat transfer in that important region. The frictional pressure drop in a tube under two-phase flow conditions can be estimated with the aid of idealized mathematical models if the phases are distributed in simple patterns. Fluid dynamical analyses have been made for constrained and unconstrained helical flow in tubes. The mathematical heat transfer and fluid dynamics models have been used to predict performance of helical single and two-phase flow systems. These results are in general agreement with the experimental behavior.

It was found experimentally that the ratio of Stanton modulus to the friction factor for helical single and two-phase flow was not a unique function as in the case of linear single-phase flow. It is believed that a wide range of ratios are possible depending upon the character of the velocity structure of the rotational flow, the phase distribution, turbulence promotion and the wall heat addition. Some aspects of the variation in the heat transfer to pumping power ratio can be explained analytically.

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CRITICAL FLOW PHENOMENA IN TWO-PHASE MIXTURES AND THEIR RELATIONSHIPS TO REACTOR SAFETY

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The expulsion of the coolant from pipe and vessel breaks in high pressure water systems will involve metastable and two-phase flow phenomena including critical or choking flows. Containment and emergency cooling systems must be designed to cope with the transient and maximum pressure buildups and with the rate of coolant loss.

Flow rates calculated for the coolant expulsion with no liquid vaporizing are significantly greater than the rates calculated for two-phase homogeneous flow. Thus a variety of analytical models has evolved to predict critical, two-phase flow through apertures including orifices, nozzles, and pipes. It is recognized, too, that the nature and positioning of the break will influence the type of coolant outflow and consequently the choice of the model. Critical flow rates predicted by the various models are presented for coolants such as water, cryogenic fluids, and liquid metals.

The literature on two-phase flow phenomena constitutes an important aspect of the understanding needed in evaluating the problems involved; however, only

relatively recent experiments provide the insight needed for the evaluation of practical cases. Available experimental results for critical two-phase flow in pipes are presented and compared to the predictions of the analytical models. The discussion includes the judgments for the merits and applicability of the models. Coolant expulsion (blowdown) tests have been carried out to establish design criteria for the prediction of the two-phase flows, and these results are also discussed and compared with the predictions.

In presenting the current status of the nuclear reactor safety programs, coolant discharge studies now in progress in the United States are briefly described. Preliminary results include investigations of the state of the fluid within the vessel (pressure and void measurements), pressure distributions in the aperture and pipe sections, variations of the L/D ratios for the discharge sections, and measurements of the voids at the critical discharge position. Nearly all tests in progress involve water as the fluid with system pressures extending up to 2000 psig. The further implementation of the testing programs with the use of cryogenic fluids provides information on the two-phase critical flow phenomena as related to nuclear space applications. The use of freon provides both visual and quantitative results.

For any given set of conditions describing a loss of coolant accident, unresolved problems still remaining for predicting the discharge rates are noted, with some estimation of the bounds for the errors. A program designed to demonstrate the integrated effects of a loss of coolant accident is described.

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Abstract

OXIDE FUEL FABRICATION AND PERFORMANCE

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In the past five years, there has been a systematic approach in the United States to design formulation, fabrication development, and performance testing of uranium oxide type fuels primarily for water and steam reactors. A basic effort has been carried out in determining physical, mechanical, and corrosion properties under irradiation and coolant environment conditions of the individual components of the fuel. Simultaneously, extensive in-reactor testing at various thermal and life conditions has provided significant operating and engineering test data on many fuel types. These data allow reliable quantitative evaluation of oxide fuels aimed at economic central station power applications. In all studies undertaken, fuel performance information has been preceded by the development of fabrication techniques suited to specific geometries. In general, fabrication techniques were aimed toward to improvement of thermal output, increase of life span of a given core, minimization of parasitic neutron absorption, and reduction of fabrication costs.

Fabrication processes include ceramic grade pelleted fuel (93-97% theoretical density) and high density compactible powder, either vibratorily compacted (83-89% T. D.) or followed by cold working (to 92% T. D.) or hot working (to 94% T. D.). The claddings used for water and/or steam applications include zirconium alloys, austenitic stainless steels, and, more recently, nickel alloys.

Statistically significant numbers of fuel elements have been irradiated to peak burnups ranging up to 35×10^{20} fissions/cc and at maximum heat flux values up to 700 000 Btu/hr.-sq. ft. The studies included water and steam environments with operation in the 550-650F temperature range, and also superheated steam environment with operation in the 1200-1300F clad temperature range.

Zircaloy - 2 clad oxide fuel has shown generally favorable performance at the conditions listed. The encouraging early results with Incoloy as a superheat fuel clad has initiated interest in it also for lower temperature steam environments.

Austenitic 18 Cr - 8 Ni stainless steel clad fuel, operating under stressed conditions, exhibited some failures in in-reactor tests after about 1.6 to 3.7 x 10^{20} fissions/cc peak burnup. The metal attack has been of an intergranular nature with the material either in the unsensitized state (water coolant) or sensitized condition (superheated steam). Cracking has occurred in the regions of highest thermal output, indicating a stress and environmental dependency.

Two other important fuel performance trends have been evidenced: a) Intentional and service defects in the clad exposing the UO_2 to the coolant indicate a much lower release of radioactivity into the coolant environment than had originally been expected. Once the faulty fuel is removed, there appears to be negligible residual activity carried in the coolant. b) Some 0.5"OD Zr-2 clad uranium oxide rod fuel has been operated at a heat flux of about 1 400 000 Btu/hr-ft^2 for about 3 x 10^{20} fissions/cc maximum burnup. With about 70% of the UO_2 molten at the areas of maximum thermal output, there is no evidence of clad deterioration as long as adequate heat transfer exists in the coolant.

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CARBIDE FUEL FABRICATION AND PERFORMANCE

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Development of massive uranium monocarbide as a nuclear fuel has included tests of several preparative methods; investigation of the binary composition range from "hypo" (less than the stoichiometric 4.8 wt % C), to greater carbon contents ("hyper"); and exploration of the effects of additives and the properties of ternary systems. Various fuel-element designs have been tested both in and out of pile. These were made with and without a liquid-metal thermal bond, in a variety of cladding compositions - e.g., type 304 stainless steel, Nb/1% Zr, and tungsten. The greatest application thus far has been in sodium cooled reactors. Model elements have reached 5000 MWd/tonne burnup in the Sodium Reactor Experiment (SRE). Ten full-scale UC elements have been operating since September 1963 in the first core loading of the 81-MW(e) Hallam Nuclear Power Facility (HNPF).

All the above fuel elements are clusters of long rods, each made up of UC slugs, thermally bonded by sodium to thin-wall cladding of type 304 stainless steel. The fuel is arc-melted and cast hypo-UC (4.4 - 4.8). A typical heat of enriched material starts with a total charge of 50 to 65 kg and results in a pour of 20 to 25 kg, which can produce about 50 slugs, 0.5 in diameter (1.27 cm) x 6 in. long (15 cm). As of January, 1964 more than 12 000 kg of slugs had been prepared in this manner, including 2000 kg of enriched material. Over-all process yields indicate that with enrichment near 3 wt % U-235, fabrication cost of complete fuel elements would not exceed $100/kg of contained U.

Irradiation behavior of fuel elements made with hypo-UC, sodium bonded to stainless steel cladding, is projected to burnup of 25 000 MWd/tonne at limitations now estimated as follows:

Carbon range:	4.5 to 4.8 wt %
Maximum fuel center temperature:	1800F (980C)
Maximum cladding temperature:	1250F (675C)

These conditions permit linear heat rates up to 26 kW/ft (865w/cm), and specific power of 50 kW/kg U in practical fuel element designs. Allowance may be made readily to accommodate anticipated dimensional chances and fission-gas release. Higher temperature performance appears possible with a narrower range of carbon content about stoichiometric and with a more refractory metal cladding.

In preparing carbide fuel, powder-metallurgy techniques are feasible if the dimensions are too small for casting, if a controlled density is required, or if solid solution alloys are of interest. Bodies with more than 95% of theoretical density have been reproducibly made by hot isostatic pressing of crushed arc-cast UC in niobium containers (10 000 lb/in.^{2} -- (7 kg/mm^2) -- for sh at temperatures between 1500 and 1600C). A comparable product is obtained by cold pressing of powders and binder at 50 ton/in.^{2} (70 kg/mm^2), followed by sintering at 1800C. Also, spherical carbide alloy particles have been prepared in a high intensity arc. Irradiation tests have been made to evaluate composition and fabrication process effects.

Thermochemical data and the results of experimental determinations are available to establish maximum operating temperatures at which UC and a number of potential cladding materials are compatible.

Studies of ternary systems have resulted in proposed isothermal sections of the following phase diagrams: U-C-W; U-C-Mo; U-C-Re ; and U-C-Nb. A complete range of solid solutions has been found to exist between UC and UN; and the existence has been indicated of solid solutions of U(C,N,O).

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Abstract

CERAMIC-COATED PARTICLE NUCLEAR FUELS

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Ceramic-coated particle nuclear fuels consist of small particles of the fuel material, each coated with a thin layer of an impermeable pyrolytic carbon or other ceramic material, usually dispersed in a graphite or other ceramic matrix. The coatings protect the fuel from chemical reactions, protect the matrix from fission recoils, and provide improved fission-product retention at a high temperature. These fuel materials are of particular interest in gas-cooled reactors.

The steps in the fabrication of coated-particle fuel materials are the fabrication of the fuel particles, their coating with a ceramic layer, and the fabrication of a dispersion of the particles.

Methods capable of producing nearly spherical fuel particles have been emphasized in the work so far. Oxide particles have been produced by sintering methods and the sol-gel process. Carbide particles have been produced by fusion, and by a modification of the sol-gel process.

Fuel particles have been coated with many ceramic materials. The method most commonly used is vapor-phase coating in a fluidized bed. The conditions have been worked out for producing strong, impermeable coatings of carbon, beryllia and alumina, and information obtained on some of the important properties of the coatings.

Gentle methods are required to fabricate ceramic dispersions of these coated particles to avoid cracking of the coatings. Certain fabrication methods have been found to be particularly successful.

Laboratory evaluations of fission-product retention show that losses of the fission gases krypton and xenon are negligible from uncracked coated particles at all practical operating temperatures. The loss of some solid fission products through pyrolytic-carbon coatings is greater than that of the fission gases.

Radiation studies have been directed towards achieving optimum fission-product retention through designing the fuel material so that cracking of coatings does not occur. It has been found that, for both pyrolytic carbon and oxide coatings, a certain minimum coating thickness is required to prevent cracking. Porosity in the fuel particle or the coating is beneficial in preventing cracking and, in the case of pyrolytic carbon coatings, a multilayer coating has also been found desirable. Recent radiation results show that essentially all coating cracking can be prevented with pyrolytic-carbon- or alumina-coated particles, to burnup of 10-20% of heavy-metal atoms, at temperatures up to 1400C. It is more difficult to obtain good performance from BeO-coated fuel materials; ways to make better BeO-coated fuels are being studied.

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URANIUM/PLUTONIUM CERAMIC FUELS

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Uranium/plutonium ceramic fuel fabrication and irradiation studies for both thermal and fast reactor applications have been accelerated rapidly in recent years. The quantity of such fuel irradiated in the United States during 1963 was twice that tested in 1962 and the amount is expected to double again in 1964.

It has been demonstrated that plutonium-bearing fuel elements can be fabricated on a pilot plant scale, safely and at little additional cost over that of fabricating uranium elements. For example, the cost of fabricating UO_2/1 wt% PuO_2 fuel elements is less than 10% greater than for UO_2. For reactors in which plutonium has unique advantages, therefore, it no longer appears logical to continue arguments that the plutonium must always be burned in situ or stored indefinitely.

Fabrication technology is being developed at several sites according to their specific interests. Processes applied to UO_2/PuO_2 include co-precipitation, mechanical blending, pressing and sintering, swaging, vibrational compaction, and high-energy impaction. The last mentioned technique provides the latest improved method for forming the most dense mixed oxides, other compounds and cermets of known composition. Sintering as well as fusion and casting of UC/PuC is being developed.

Demonstration-scale production and testing of mixed-oxide fuel elements has highlighted problems associated with use of conventional industrial equipment. In some instances impurities introduced into the fuel processing stream by such equipment caused fuel element failures. Fuel holdup, loss, and criticality control measures are now well developed.

Increased attention is being directed to adapting the newer fabrication techniques to partially decontaminated, irradiated uranium/plutonium fuels and to developing novel fuel elements such as those that can be repeatedly re-enriched, or rejuvenated and reirradiated without full chemical reprocessing. Beyond this, concepts for extending the useful life of a fuel element, such as by alternating it between thermal and fast reactors, are under investigation. Preliminary results are promising.

Although studies of compounds other than oxides have been fruitful, data from fabrication, irradiation, and fundamental studies of oxides have been developed more rapidly. For example, tons of mixed plutonium/uranium oxide were irradiated during the past year, compared to tens of kilograms of mixed carbides and a few kilograms, at most, of other compounds.

The effects of plutonium additions to the properties of uranium compounds are partly known, and uranium/plutonium oxides generally appear capable of adequate performance in thermal or fast reactor power stations now being designed. Plutonium oxide has been irradiated to exposures of more than 66×10^{20} fissions/cm^3 (200 000 MWd/ton) at heat generation rates exceeding those anticipated in fast reactors. UO_2/20 wt% PuO_2 has been successfully irradiated to 33×10^{20} fissions/cm^3 (100 000 MWd/ton) and important information on fission-gas release and fuel relocation obtained. Transient tests to several hundred times normal operating powers were conducted. Temperature-modified reactivity effects, fuel segregation, fission-product movement and melt-down effects need further study.

Irradiation tests of mixed uranium/plutonium carbide have been conducted successfully by several sites in the United States, some to more than 13×10^{20} fissions/cm^3 (43,600 MWd/ton). Other current tests have goal exposures exceeding 27×10^{20} fissions/cm^3 (91 000 Mwd/ton). These fuel experiments in support of sodium-cooled reactor technology are being supplemented by studies of physical and thermal properties, and of compatibilities with cladding materials.

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Abstract

THE SOL-GEL PROCESS FOR THE REMOTE PREPARATION AND FABRICATION OF RECYCLE FUELS

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The sol-gel process converts the nitrate solution products from solvent-extraction processing of spent reactor fuels to fuel materials suitable for use in remote fuel fabriaction in four simple and economical steps. These steps are: denitration to obtain a reactive oxide; dispersion of this oxide in dilute nitric acid to form a hydrosol; gellation by evaporation or extraction of water from the sol; and calcination to obtain the final ceramic form desired. The process can produce either the oxides or carbides of thorium, uranium, or plutonium. It can produce powder suitable for vibratory packing into metal tubes, or microspheres for subsequent coating and use in high-temperature gas-cooled reactors. Because of this flexibility, the sol-gel process promises to become an almost universal method of preparing ceramic reactor fuel materials.

The most advanced application in our development program is the preparation of mixed thorium/uranium-233 oxide for loading into fuel tubes by vibratory compaction. A ton of this fuel has been prepared and loaded into 1000 zircaloy tubes, 40-in. (101.6 cm) long and 1/2-in. (1.27 cm) diameter, in the Kilorod Facility at Oak Ridge National Laboratory. This facility includes shielded and remotely operable equipment for the preparation of dense particles of mixed ThO_2/UO_2 by the sol-gel process, conditioning of the powder by grinding and sorting by size vibratory-compaction loading of the Zircaloy tubes, and welding and inspection of the completed element. After correction of initial mechanical difficulties, operation at 90% of the design capacity was routinely

attained with less than 3% of the elements requiring recycle because of density defects. This facility, which is our first remotely operable, integrated fuel preparation and fabrication facility, is the forerunner of the Thorium-Uranium Fuel Cycle Facility now being designed for future development of integrated fuel recycle processes.

From analysis of the radiation levels observed in the Kilorod Facility, we conclude that on a 10-kg batch scale, reactor fuel rods containing U-233 contaminated with about 500 parts/million of U-232 can be safely fabricated by use of the shadow shielding provided by about a 2-in. thickness of lead. However, final assembly of the fuel bundle, larger-scale operation, or higher levels of U-232 contamination will require complete shielding and remote operation.

An irradiation testing program is being conducted on vibratorily compacted, sol-gel thorium oxide fuels. Our findings indicate that this is a suitable fuel form for power reactor application. Irradiations to 35 000 MWd/ton at peak heat ratings of an $\int kd\text{-}1$ of 40 watts/cm have produced insignificant changes in the oxide. In capsules now being irradiated, burnups have reached 75 000 MWd/ton. At heat rating of an $\int kd\text{-}1$ of 60 watts/cm, recrystallization and sintering of the oxide are observed, but with no noticeable deleterious effects. Fission-gas release from the sol-gel oxide is comparable to that from fused oxide.

By introducing fine-particle-size carbon into the oxide sol, a mixed oxide-carbon gel can be prepared. After drying, these gels can be reacted in argon at about 1750C to give high-purity carbide. Mixed thorium-uranium dicarbide has been prepared in this manner with a density of 8.9 g/cm^3 and sufficient purity for use as a reactor fuel. For the preparation of UO_2 or UC_2, it is necessary to keep the uranium mostly in the tetravalent state. Uranium dioxide sols satisfactory for the process have been prepared in two ways: by precipitation of $U(OH)_4$ from uranous nitrate solutions followed by aging to obtain a concentrated sol, and by electromigration of the uranyl ion into the cathode compartment of an ion-exchange membrane cell containing very dilute nitric acid. At the cathode, the uranium is reduced to the tetravalent state and precipitates to produce a sol that can be concentrated by evaporation.

resist the swelling of the fuel, irradiation damage resistance, and attractive nuclear properties. Vanadium-base alloys are preferred for this application although niobium-base alloys and duplex tubing are of interest. Fabrication of high-quality small-diameter tubing of these more exotic jacket materials has been difficult. Disposable-powder core extrusion and ductile core-drawing techniques being developed have been studied, along with mechanical properties, compatibility, and corrosion resistance of these alloys.

$(U,Pu)O_2$ fuel is amenable to several fabrication techniques. Pressing and sintering of high-density pellets, powder filling and swaging of high-density rod, and vibratory compaction of graded powders are methods of producing rod-type fuel elements of the mixed oxide. These methods have been studied in detail, as have their adaptability for remote operation from the standpoint of dealing with high Pu-240 concentrations. The physical properties of the $(U,Pu)O_2$ fuel are important. Stainless steels are attractive jacket materials for $(U,Pu)O_2$; other high-strength alloys for containing released fission gases, however, could also be of interest.

(U,Pu)C fuel is adaptable to the same ceramic fabrication techniques as the oxide. Increased chemical activity of this material over the oxide requires careful atmospheric control. The methods for synthesis of the carbide include direct reaction of molten plutonium and graphite. Vibratory compaction of powders is an important method of fabrication particularly when remote operation is considered, although pressing and sintering are important in producing high-density bodies. The use of stainless steel as a jacket material for (U,Pu)C is desirable from many standpoints. This material, however, may not satisfy compatibility requirements. Alternative jacket materials such as vanadium alloys and duplex tubing utilized for metallic fuels are applicable here.

Cermet fuels, using UO_2 or PuO_2 as the dispersed fuel phase, have potential for fast-reactor application. Where high breeding ratio is desired, uranium or thorium alloy can be used as the fertile matrix material; but where it is not, stainless steel can be used as a nonfertile matrix. Methods have been investigated for the fabrication of pin and plate elements.

Thorium-base alloys containing uranium have exhibited potential as fuel. This fuel is amenable to fabrication by casting into small-diameter rods and by working methods such as extrusion, swaging and rolling. Thorium/Plutonium and Th/U/Pu alloys also have potential as fast fuels.

In all cases, the physical and mechanical properties of both fuel and cladding are important in the fabrication processes and economics.

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Abstract

IRRADIATION BEHAVIOR OF METALLIC FUELS

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Alloys containing uranium, plutonium, and thorium have been investigated extensively in order to develop metallic fuels with improved properties under irradiation. Several fuel alloy systems have been developed which show excellent dimensional stability under irradiation. Anisotropic irradiation growth of alpha uranium can be virtually eliminated by addition of alloying elements such as molybdenum, niobium, and zirconium. A suitable heat treatment, usually consisting of an extended anneal just below the upper temperature limit of the alpha phase, is required if the material has been fabricated with a preferred grain orientation.

High temperature swelling in uranium fuels can be reduced by alloying additions, again often in connection with a suitable heat treatment to produce an optimum form of microstructure. Swelling in metastable gamma phase alloys in some cases is less dependent upon the phases present before irradiation, but at temperatures below the stable gamma phase the amount of swelling appears to be significantly influenced by fission rate. Swelling in metallic fuels can also be minimized through use of cladding restraint. For example, cladding thicknesses on the order of 0.25 mm have been found to be capable of raising the allowable operating temperatures and achievable burnups in uranium-

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Abstract

IRRADIATION BEHAVIOR OF CERAMIC FUELS

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A review is presented of the extensive investigations made since 1958 of various ceramic compounds of uranium, plutonium, and thorium for use as reactor fuel materials. The ceramic fuel materials most extensively investigated are the uranium oxides, followed by the carbides and to a lesser extent the nitrides and the silicides. This paper reviews the effects of radiation at various power levels on the properties and performance characteristics of ceramic fuel materials.

The ease of fabricability, stability under irradiation, and compatibility with high temperature water coolant has led to the successful use of the oxides, principally UO_2 and to a lesser extent ThO_2-UO_2 and PuO_2-UO_2, as fuel materials for water cooled power reactor applications. Other oxides being investigated include ZrO_2 mixtures with UO_2 and PuO_2, BeO-UO_2, UO_2-CaO-ZrO_2, UO_2-CeO_2 and Mgo-PuO_2.

UO_2 pellet and compacted powder fuel rods have been operated at thermal ratings high enough to result in melting of the central portion of the fuel. Observations of volume swelling of fuel rods operating with molten UO_2 and the use of cored pellets or lower density compacted UO_2, to eliminate volume swelling are discussed.

The results of UO_2 central melting experiments which have led to improved correlations between fuel rod thermal ratings, UO_2 thermal conductivity, changes in UO_2 structure, and identification of molten zone are discussed. Data are presented on the thermal conductivity of polycrystalline as-sintered

UO_2 at elevated temperatures and on changes in thermal conductivity of UO_2 with burnup.

Quantitative data are presented on the volume swelling rate of UO_2 as a function of burnup, UO_2 density, and operating temperature. Other significant data on UO_2 fuel include fission gas release measurements on fuel operating up to and above the melting temperature, observation of metallic inclusions of uranium in UO_2 irradiated at high temperatures, and low-temperature sintering of compacted powder UO_2 during irradiation.

The results of extensive irradiation tests of other oxide fuel materials such as ThO_2-UO_2, PuO_2-UO_2, and ZrO_2-UO_2 are presented. Results of recent tests of PuO_2-UO_2 fuel irradiated to 20 x 10^{20} fissions/cc indicate that no radial migration of Pu, Zr-95 or Ce-144 occurred. ZrO_2-UO_2 was found to be a stable material for use in plate-type elements. The swelling rate of ZrO_2-UO_2 with burnup was measured to be the same as that of UO_2.

The higher thermal conductivity and metal atom density of the carbides relative to the oxides has provided an incentive for the investigation of such ceramic fuel compounds as UC, PuC and UC-PuC mixtures. The most extensively studied of the carbides is UC. Data on the relative stabilities of hypo-stoichiometric and hyperstoichiometric UC as a function of operating temperature and irradiation exposure are presented. Peformance data for PuC and UC-20 w/o PuC, irradiated to burnups of 7.1 x 10^{20} fissions/cc, are discussed.

Uranium monoitride and uranium monosulfide have been investigated as potentially better ceramic fuel materials than UO_2 and UC. The results of irradiation tests of UN and US are reviewed.

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Abstract

IRRADIATION BEHAVIOR OF FLUID FUELS

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Three types of fluid fuels have been studied in the United States in the past five years. These include aqueous solutions and slurries, solutions of uranium or plutonium in liquid metals, and solutions of uranium tetrafluoride in molten fluorides.

Operation of Homogeneous Reactor Experiment-2 (uranyl sulfate solution) demonstrated safe operation of a fluid-fuel reactor, maintenance of a highly radioactive system, and recombination of radiolytic H_2 and O_2 by a homogeneous catalyst Cu^{++}. However, power-dependent deposition of uranium on the Zircaloy-2 core vessel was encountered with resultant severe corrosion and melting of the metal. Improvement of the core hydrodynamics apparently obviated the deposition under HRE-2 conditions; it was concluded a suitably designed system would be operable, but fuel and material requirements leave little margin for error.

Operation of LAPRE-II showed the fuel, a solution of uranium oxide in phosphoric acid, to be stable under irradiation and to recombine the H_2 and O_2 effectively without added catalyst; practicable materials of construction have not been demonstrated.

Aqueous oxide slurries have also been explored, but most of the effort in the U. S. was devoted to thoria slurries for blankets. Out-of-pile studies showed

that flocculated thoria suspensions prepared with 1 to 5 μm attrition-resistant particles had practicable engineering properties; however, in-pile experiments showed such particles underwent extensive comminution under reactor (fissioning) conditions. No radiation enhancement of corrosion was observed.

A sodium-cooled reactor (Los Alamos Molten Plutonium Reactor Experiment), whose fuel is a stagnant molten alloy of plutonium and iron, was operated with the tantalum-capsule/fuel interface as hot as 690C. Penetration of the weld regions and some thinning of the capsule walls at the liquid/vapor interface occurred both in- and out-of-pile. A less corrosive plutonium/cobalt/cerium alloy will be used in the near future. About 95% of the fission gases are evolved from the fuel, but a progressive reactivity loss due to growth of gas bubbles on the capsule walls is observed during operation.

The Liquid Metal Fueled Reactor, a graphite-moderated reactor fueled with a circulating, dilute solution of uranium, zirconium and magnesium in bismuth, was stopped short of completion. Testing in a circulating loop under irradiation showed the fuel solution to be stable and suggested no evidence of enhanced corrosion.

The graphite-moderated MSRE will be fueled with a solution of UF_4 in a melt of LiF, BeF_2 and ZrF_4. Graphite, metal and fluoride are completely compatible in pre-irradiation tests. Under irradiation at proposed operating temperatures (650C) very small quantities of CF_4 are formed, but corrosion of the metal remains negligible. The graphite remains unwetted and essentially undamaged. Fluorine is liberated from such fuels by beta and gamma radiation below about 150C, and some of the observed effects are attributable to this phenomenon. Most fission products remain dissolved as fluorides, noble metals may deposit from the melt as metals or carbides, and gases are evolved from the fuel and partially retained by the porous moderator.

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Abstract

RADIATION-INDUCED CHANGES IN THE PROPERTIES OF NON-FUEL REACTOR MATERIALS

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The engineering properties of nuclear structural metals, including many properties that dictate the useful longevity of reactor components, have been shown to undergo drastic changes during exposure to neutron irradiation. Extensive investigations of these changes have been made in recent years. As a result, considerable understanding of observed property changes in terms of environmental parameters has been accumulated. Such understanding and its application through the formulation of radiation damage models are important for the prediction of safe material operating limits in nuclear reactors.

An increasing awareness of the roles played by the temperature at which the irradiation exposure is accumulated and the alloy content of the metal under irradiation has developed in recent years. Long-time properties as determined by in-reactor tube burst tests and postirradiation stress-to-rupture tests of certain high temperature alloys have been shown to be markedly reduced by both low temperature and high temperature irradiation exposures. While postirradiation annealing has been shown to be quite effective in eliminating transition temperature effects in ferritic steels, it has been impossible to produce recovery of irradiation effects on stress-to-rupture properties by annealing in a number of precipitation-hardening high temperature alloys.

Short-time tensile properties of Zircaloy-2, the austenitic stainless steels, and nickel-base alloys show a varied response to irradiation and irradiation temperature. In both Zircaloy-2 and the austenitic stainless steel, considerably greater room temperature strength increases and ductility reductions occur at an irradiation temperature of 290C than during similar irradiation exposure at 60C. High temperature irradiations of a precipitation-

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CORROSION IN AQUEOUS SYSTEMS

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The successful operation of water-cooled power reactors has established the good corrosion resistance and overall satisfactory performance of Zircaloy-2 and stainless steel as fuel element jacketing materials.

In some experiments, Zircaloy-2 has been shown to exhibit multiple transitions in corrosion rate in high temperature water; it is inferred that periodic breakdown in the protectiveness of the oxide occurs. After exposure in pressurized water reactors, boiling water reactors and in-reactor loops, Zircaloy-2 jacketing has sometimes shown no effect of reactor environment on corrosion; at others there has been a significant increase.

For water exposure at normal reactor temperatures, only the reduction in mechanical properties, as a consequence of hydrogen absorption, appears to be a potential limitation to the useful life of the alloy. If the temperature is increased above the range 400-450C, the oxide film loses its tenacity and the corrosion rate increases substantially. Thus, extrapolation of the use of Zircaloy to high temperature superheated steam applications does not seem feasible.

A modification of Zircaloy-2, designated Zircaloy-4, has retained the favorable properties of Zircaloy-2 and has reduced hydrogen pickup during water corrosion. Other zirconium alloys have improved resistance to corrosion by superheated steam. At temperatures as high as 650C hydrogen absorption is slight, but embrittlement due to oxygen pickup is potentially serious. Also, niobium alloys have been developed which have higher strength and equivalent resistance to water corrosion; in superheated steam, certain niobium alloys have lower corrosion rates than do zirconium alloys.

The corrosion rate of chromium-nickel stainless steels is low in water at temperatures of interest. In superheated steam corrosion rates of austenitic

steels increase markedly above 540C. Both from the point of view of penetration and of corrosion product formation, rates are probably unacceptably high for type 304 stainless steel at 650C. Certain chromium-bearing ferritic steels have shown substantially better corrosion resistance at this temperature than the austenitc steels. Some nickel-base alloys also exhibited the lower corrosion rates.

For stainless steels, the problem of major concern has been stress-corrosion cracking. It has been observed in heat exchangers; in superheated steam, where a number of experimental stainless steel jacketed fuel elements failed; in reactor water at a wide range of temperatures; in high temperature uranyl sulfate solution. The presence of chloride ion has generally been indicated to be necessary for cracking to occur. In low chloride environments, local concentration of the ion is observed as a result of evaporation with limited mixing, and of the ion exchange characteristics of deposited oxides. Certain of the nickel-base alloys seem immune to chloride-induced stress-corrosion cracking.

Susceptibility to stress-corrosion cracking has been related to the stress-induced formation of a parallel network of dislocations, and to the "mean atomic diameter" of metal atoms in nickel-chromium-iron alloys.

In nearly pure water, the corrosion of carbon steels and low alloy steels is accelerated by minor amounts of oxygen dissolved in the water. At higher oxygen contents, depending on temperature, protective films are formed and corrosion rates are again low.

In pressurized water systems, the pH is controlled in the alkaline range to minimize deposition on heat transfer surfaces of transported corrosion products. However, alkali metal hydroxide content increases at crevices where local boiling occurs and where alternate wetting and drying occur in steam phase areas. Such strong solutions can induce stress-corrosion cracking in carbon and stainless steels, and cause marked acceleration of the corrosion of Zircaloy-2 and low alloy steels. In the latter case, lithium hydroxide is less dangerous than the other alkalis, apparently because its concentration is prevented from reaching a harmful level by the formation of a mixed lithium iron oxide.

In aqueous systems, corrosion products are removed from the site of formation by dissolution as well as by physical removal of particles. The particulate material deposits at places of change in flow direction, in turbulent regions, on heated surfaces and in radioactive areas. Dissolved oxides do not generally deposit at such locations, but precipitate where the temperature is low.

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CORROSION IN LIQUID METAL SYSTEMS

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Liquid metal corrosion investigations have been stimulated in the last few years by the development of fast reactors and the interest in space power. In the latter applications, liquid metals are considered for use not only in reactors but also in power conversion systems. This paper presents corrosion data obtained in liquid and boiling liquid metal systems contained both in conventional materials of construction and in the refractory alloys.

Although there has been a general increase in the understanding of corrosion mechanisms the most important advance in recent years has been the recognition of the role of O_2, C, N_2, H_2 in corrosion by liquid metal coolants. In the liquid alkalis these elements often cause the observed physical property changes and corrosion. Analytical techniques for the determination of these elements in liquid metal systems have been improved to permit careful study of their effects. Also the requirements of environmental control to minimize contamination of refractory metals have been defined.

The liquid metals currently being studied are Li, Na, K, Rb, Cs, Hg, and the Pu 2.5 w/o Fe eutectic.

Lithium leaches Ni and most non-metallics from its containment at high temperatures. Attack of refractory alloys is highly sensitive to O activity in the solid, but not in the liquid metal: Cb-1Zr resists the attack observed on unalloyed Cb. Nitrogen in Li increases the solubility, and therefore the rate of attack of Fe, Cr, Ni, Ti, Mo, and Cb.

Sodium can be contained at temperatures up to 600C in the austenitic stainless steels; above this temperature Ni and chromium are leached. Na has been contained in Cb-1Zr up to 1200C. Above 600C Cb-1Zr removes oxygen from Na containing more than 5 ppm oxygen and is embrittled. Although carbon transport by Na is well known, its complex chemistry is only now being understood.

Potassium is less aggressive to austenitic stainless steels than sodium at temperatures up to 850C. At 1000C Cr and C co-transfer; however nickel is not leached. Hot trapped K does not attack Cb-1Zr up to 1100C. However, the behavior of oxygen in K is more complex than in Na.

Rubidium tends to leach Zr from Cb-1Zr and Ti from Mo-0.5% Ti at 1100C. At 925C these alloys as well as V and Haynes-25 are not attacked. Oxygen accelerates attack by spinel formation.

Isothermal transfer through Cs of both interstitial and metallic constituents of refractory alloys has been observed at temperatures above 1000C.

Ferritic steels and Haynes-25 corrode at low rates in boiling Hg systems at 600-650C. Zirconium is slightly better than Ti as a dissolved corrosion inhibitor. Mercury leaches Ni from austenitic stainless steels. Columbium is corrosion resistant to Hg; cold worked Cb-Zr alloys are susceptible to embrittlement at temperatures above 600C.

Plutonium-iron eutectic reacts with interstitial impurities in Ta capsules at temperatures $\geq$ 625C, resulting in a penetration of the Ta similar to the attack by Li on oxygen-contaminated Cb.

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NONDESTRUCTIVE TESTING OF NUCLEAR REACTOR COMPONENTS

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Design criteria, materials, and fabricating processes for nuclear reactor components are not sufficiently perfect to guarantee the high standards of quality required for functional and safety reasons. Even with the best manufacturing processes, properly controlled and executed, there are inherent flaws. Nondestructive testing provides a way of detecting the presence of inhomogeneities in materials without damage to the material.

During the past several years much information on nondestructive testing of nuclear reactor components has been accumulated. The research and development on nondestructive testing has provided assistance for improving the quality and reliability of reactor components. As a result of this research and development program, improvements have been made in old methods and techniques; new methods and techniques have been developed, and specialized instrumentation has been evolved. The sensitivity and resolution of all techniques are continually being improved.

Nondestructive testing of components for reactor systems differs from ordinary industrial nondestructive testing because the specifications for the nuclear systems are more rigid than those required in most other industries. The sensitivity and resolution required of the tests are in excess of those that can be justified on a cost basis in most industrial applications.

Detecting flaws and measuring material properties are among the useful applications of nondestructive testing. Requirements for quality work extends from the raw material through the completely fabricated component. One of the most valuable aspects of nondestructive testing is its effect on maintaining the quality of fabricated components and improving processes.

This paper summarizes the significant research and development in nondestructive testing since 1958 by various U. S. Atomic Energy Commission sites and contractors. The paper will include the application of nondestructive testing to fuel elements, tubing, piping, and pressure vessels. The methods to be discussed include acoustics (sonics and ultrasonics), penetrants, penetrating radiation (X-rays, gamma rays, beta rays and neutrons), eddy currents, leak, and thermal. Specific problems to be discussed include bond integrity, clad integrity and thickness, grain size and orientation, weld integrity, Uranium 235 content, fuel homogeneity, and component spacing. The special techniques to be described include Lamb waves, neutron radiography, scintillation spectrometry, high and low voltage radiography, remote inspection, backscatter laminography, pulsed eddy currents, and ultrasonic imaging. The advantages and limitations of and standards for the various methods and techniques will be evaluated.

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Abstract

PLUTONIUM VALUE ANALYSIS

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Several definitions used to estimate the value of plutonium as thermal reactor fuel are discussed and careful distinction is made from pricing, a complex subject involving considerable speculation beyond the scope of this work. Most analyses determine as "value" the price of plutonium for which the reactor fuel costs for each of two alternative fuels are the same. Limited insight can be gained from such values because, in practice, many alternatives are often available. Even further simplified formulations of costs and business strategies have been used to ascertain the impact of only one variable. Results of such brief analyses are often inconsistent with results of other analyses. Many of these inconsistencies are eliminated, or at least the reasons for the inconsistencies are less obscure, if one adheres to comparing costs at defined operating points considering sale and purchase of marketable materials. This methodical approach appears tedious and irritating to the physical scientist, but preserves sound business logic.

The various methods to determine the value of plutonium as a power reactor fuel are classified as practical and as abstract, and selected results are compared. Included in the practical methods are computed plutonium values for specific reactor types and economic environments; while the abstract methods involve models designed to delineate and to classify particular effects observed from "practical" computations. At this time, even "practical" calculations require assumptions and idealizations to allow solution; consequently, interpretation of the results is particularly dependent upon knowing the limitations of the value definitions.

Results of a "practical" analysis of plutonium as a successively recycled fuel for several simulated thermal reactor systems are presented and

correlation of the resulting variations of plutonium values with the isotopic composition of plutonium is described. Because consideration of successive recycle without interchange of fuel between different reactors is possibly unreal, methods of treating this larger problem are discussed.

Generally speaking, calculation of plutonium values is somewhat premature because numerous facts are insufficiently known at this time. Many of the uncertainties are compensative if plutonium values are determined by comparison with slightly enriched uranium cycles as such cycles burn plutonium in situ. However, in other important calculations, errors in key factors are cumulative. Computed examples are shown.

Uncertainties due to the inherent nature of each plutonium isotope are illustrated, such as the degree of self-shielding of the absorption resonance of plutonium-240. Less generally known, but important, is the degree of self-shielding achieved for the plutonium-239 resonances, which heavily influence the alpha value of plutonium-239.

The limitations to simplified plutonium value analysis increase as the complexity of the nuclear system considered increases. The use of plutonium to enrich thorium and of uranium-233 to enrich uranium-238 are examples, as is exchange of plutonium fuel between fast and thermal reactors. The influence of a firm market for decay heat sources, such as curium-244 and plutonium-238 (albeit an uncertain market) upon the design of thermal reactor fuel cycles is discussed.

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Abstract

FUEL CYCLE ECONOMICS OF URANIUM FUELED THERMAL REACTORS

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This paper discusses current and near-term technical and economic aspects of the fuel cycles of four classes of uranium fueled thermal reactors: Boiling Water (BWR), Heavy Water (HWR), Pressurized Water (PWR), and Sodium Graphite (SGR).

Thermal reactors using the thorium fuel cycle and plutonium recycle are discussed in other U.S. papers.

Actual performance data for complete irradiation cycles of the reactors under discussion herein are still somewhat limited in availability, although the performance that has been accrued is very encouraging. As of the end of 1963, three large prototype power reactors have been refueled in the U.S. One of these reactors is the Shippingport PWR which is not specifically discussed in this paper although it has provided a large amount of information on the performance of uranium fueled thermal reactors. The other two are the Dresden BWR and the Yankee PWR. In each of these cases, the fuel performance was generally as had been predicted. The fuel costs actually experienced to date are of course substantially greater than cost estimates for future fuel discharge batches.

The Hallam SGR and the Carolinas-Virginia HWR are small prototype plants, each of which is operating on its first core loading.

A number of analytical approaches can be followed in computing the fuel cost of a reactor once the fuel performance characteristics have been specified. In order to achieve a better understanding of the fuel costs discussed in this paper, a specific computation of fuel cost is set forth in detail. A method of computing "working capital" charges on the fuel cost is also presented. Depending on the overall cash flow as a function of fuel cycle

time, there can arise important differences in working capital requirements in support of fuel cycle operations for reactors which otherwise might have similar fuel costs. Private ownership of the nuclear fuel inventory is an especially important case requiring an accurate assessment of working capital requirement.

Reactors presently under construction (PWR's and BWR's) and reactors which can be constructed in the immediate future (HWR's and SGR's) have estimated fuel costs which are markedly lower than plants currently in operation. Estimates for near-term economic performance of the fuel cycle of each of the thermal reactors under discussion herein are presented and discussed.

Besides making technical progress in reactor fuel performance characteristics, improvement is being made in the unit charges of the various supporting operations for the fuel cycle. At the last Geneva Conference, the U.S. schedule of charges for enriched uranium could be found to correspond with a charge for natural uranium feed of $39.3 per Kg uranium as UF_6 and a charge for separative work of $37.3 per Kg uranium. Now, the U.S. schedule of charges for enriched uranium can be found to correspond with natural uranium feed at $23.5 per Kg uranium as UF_6 and separative work at $30 per Kg uranium. This change alone has reduced thermal reactor fuel costs by some 0.5 mil/KWH.

The change in fuel cost arising from changes in selected parameters, both of a technical and economic nature, is discussed and quantitative data for selected cases are presented.